TVA

**Tennessee Valley Authority**, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

June 5, 2003

TVA-SQN-TS-03-08 10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

In the Matter of ) Docket No. 50-328
Tennessee Valley Authority )

**SEQUOYAH NUCLEAR PLANT (SQN) - UNIT 2 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 03-08, "REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN CURVES"**

Reference: TVA letter to NRC dated September 6, 2002, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 Technical Specification (TS) Change No. 00-14, 'Pressure Temperature Limits Report (PTLR) and Request For Exemption From The Requirements Of 10 CFR 50, Appendix G' "

In accordance with 10 CFR 50.90, TVA is submitting a request for an amendment to SQN's License DPR-79 to change the TSs for Unit 2. The proposed amendment revises TS 3/4.4.9.1, "Pressure/Temperature Limits, Reactor Coolant System." The revision replaces the pressure-temperature (P-T) limits that are currently analyzed for 14.5 Effective Full Power Years (EFPYs) with new limits analyzed for 32 EFPY. In addition, the amendment includes corresponding changes to the TS figure associated with Low Temperature Over Pressure Protection (LTOP) and the TS Bases.

TVA's reference letter requests a TS amendment for SQN (both units) to incorporate a PTLR. The PTLR contains updated P-T limits applicable for 32 EFPYs. The limits are based on NRC approved methodology with two exceptions. One of the

D030

exceptions eliminates the minimum temperature requirement evaluation for the reactor pressure vessel (RPV) closure flange region. Following recent discussions with NRC staff, TVA understands that review and approval of TVA's requested change that eliminates evaluation of the flange region may not be complete before the applicability of SQN's present Unit 2 limits expire. The SQN Unit 2 P-T limits are currently applicable for 14.5 EFPYs which is projected to expire in early August 2003. Accordingly, TVA, as an interim measure, is submitting new Unit 2 P-T limits that include the minimum temperature requirement evaluation for the RPV flange closure region.

NRC approval of the new Unit 2 P-T limits will allow operation of Unit 2 beyond the August 2003 expiration date. TVA understands that NRC will continue their ongoing review of TVA's reference PTLR request. The next milestone associated with P-T limits is SQN Unit 1 with current limits applicable for 16 EFPY. The current Unit 1 limits are projected to expire in 2005.

It may be noted that the topical report that supports the enclosed amendment (Topical Report, WCAP-15321, Revision 1) utilizes alternatives to the requirements of 10 CFR 50, Appendix G as referenced by 10 CFR 50.60(a). In TVA's reference letter, TVA utilized exemptions from Appendix G. The first exemption utilized American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T limit Curves for ASME Section XI, Division 1," in lieu of 10 CFR 50, Appendix G, paragraph IV.A.2.b. The second exemption utilized WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants" in lieu of 10 CFR 50, Appendix G, footnote 2 to Table 1. The first exemption associated with Code Case N-640 is required for the enclosed amendment. Accordingly, TVA requests that NRC apply the N-640 Code Case exemption (previously submitted by TVA's reference letter) to the enclosed amendment.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The SQN Plant Operations Review Committee and the SQN Nuclear Safety Review

Board have reviewed this proposed change and determined that operation of SQN Unit 2, in accordance with the proposed change, will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Tennessee State Department of Public Health.

TVA requests approval of this TS change by August 1, 2003, and that implementation of the revised TS be within 15 days of NRC approval.

There are no regulatory commitments contained in this submittal. This letter is being sent in accordance with NRC RIS 2001-05. If you have any questions about this change, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 5 day of June, 2003.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Enclosures:
1. TVA Evaluation of the Proposed Changes
2. Proposed Technical Specifications Changes (mark-up)
3. Changes to Technical Specifications Bases Pages

cc: See page 4

Enclosures
cc (Enclosures):

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2

## 1.0 DESCRIPTION OF THE PROPOSED CHANGE

This letter is a request to amend Operating License DPR-79 for SQN Unit 2. The proposed change would revise the Operating License for SQN Unit 2 Technical Specification (TS) 3/4.4.9.1, "Pressure/Temperature Limits, Reactor Coolant System." The proposed amendment deletes two figures (Figures 3.4-2 and 3.4-3) that are referenced in Limiting Condition for Operation (LCO) 3.4.9.1. and provide pressure-temperature (P-T) limits for reactor coolant system (RCS) heatup and cooldown. The figures are currently applicable for 14.5 Effective Full Power Years (EFPYs) and are being replaced with updated figures analyzed for 32 EFPYs.

In addition, the proposed change includes a revision to TS 3/4.4.12 and the associated Figure 3.4-4 that provides low temperature overpressure protection (LTOP) setpoints. The proposed revision extends the applicability of the setpoints to 32 EFPYs and revises the Appendix G limit to reflect the updated 32 EFPY cooldown limit.

The proposed changes described above include appropriate revisions to the associated TS Bases sections.

## 2.0. PROPOSED CHANGE

In summary, TVA's proposed change updates the SQN Unit 2 RCS heatup and cooldown curves (P-T limits) and extends the life of the Unit 2 limits from 14.5 EFPYs to 32 EFPYs. In addition, the amendment includes a change to the TS figure associated with the Low Temperature Over Pressure Protection (LTOP) system and provides corresponding changes to the TS Bases.

## 3.0. BACKGROUND

By letter dated September 6, 2002, TVA requested a TS amendment to update SQN's current P-T limits and incorporate a Pressure Temperature Limits Report (PTLR) within the SQN TSs for both units. The updated P-T limits provided in the SQN PTLR contain heatup and cooldown limits applicable for 32 EFPYs. The limits are based on the latest analytical methodology with some exceptions. One exception included the elimination of the minimum

temperature requirement evaluation for the reactor pressure vessel (RPV) closure flange region. Following recent discussions with NRC staff, TVA understands that review of the change may not be complete before applicability of the present Unit 2 P-T limits expires. The Unit 2 P-T limits are currently applicable for 14.5 EFPYs and are projected to expire in early August 2003. Accordingly, TVA, as an interim measure, is submitting new Unit 2 P-T limits that account for the minimum temperature requirements for the RPV closure flange region. NRC approval of the new Unit 2 P-T limits will allow operation of Unit 2 beyond the projected August 2003 expiration date.

## 4.0 TECHNICAL ANALYSIS

Title 10 of the Code of Federal Regulations, Part 50, Appendix G, requires the establishment of P-T limits for specific material fracture toughness requirements of the reactor coolant pressure boundary materials. The 10 CFR 50, Appendix G establishes an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G.

The components of the RCS at SQN are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips.

The P-T limits, as established by the requirements of 10 CFR 50, Appendix G, are periodically reanalyzed and revised as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. The P-T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. Periodic adjustment of the P-T limits is needed to account for these time-dependent parameters. Adjustment of the limits is acceptable if performed in accordance with methodology approved by the NRC. TVA's proposed change utilizes analytical methods approved by NRC as prescribed in Westinghouse Topical Reports WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January 1996." In addition, a plant specific analyses for SQN's Unit 2 P-T limits is provided in WCAP-15321, Revision 1, "Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation, April 2001."

The updated limits are within applicable plant design assumptions as discussed in Section 5.2.4.3 of the SQN Final Safety Analysis Report (FSAR). These safety analyses demonstrate that SQN's Unit 2 reactor vessel is adequately protected against brittle fracture when operated within these limits.

Each P-T limit curve defines an acceptable region for normal plant operation. The curve is used for operational guidance during heatup and cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

TVA's proposed TS change updates current RCS P-T limit curves for Unit 2 and extends the applicability of these limits from 14.5 EFPYs to 32 EFPYs. The proposed change is based on analyses documented in WCAP-15321, Revision 1. The revised P-T limits include the minimum temperature requirements for the RPV closure flange region as prescribed in 10 CFR 50, Appendix G. The 10 CFR 50, Appendix G requirements address the metal temperature of the reactor vessel in the closure head flange and vessel flange regions. The requirement limits the normal operating metal temperature in this region to temperatures which exceed the material unirradiated nil-ductility transition reference temperature ($RT_{NDT}$) by at least 120 degrees Fahrenheit (°F) when the system pressure exceeds 20 percent of the pre-service hydrostatic test pressure. For SQN Unit 2, the limiting unirradiated $RT_{NDT}$ of -13°F occurs in the reactor vessel closure head flange. In addition, 20 percent of the pre-service hydrostatic test pressure (3106 pounds per square inch [psi]) is 621 psig. Based on this data, the minimum allowable temperature of the flange region is 107°F at pressures greater than 621 psig. When instrumentation margins of 10°F and 60 psig are considered (reference WCAP-15321, Revision 1), the allowable temperature in this region becomes 117°F at pressures greater than 561 psig. Accordingly, this limit has been included in the revised Unit 2 P-T limit curves.

The LTOP system setpoints and the RCS vent size are evaluated for compliance each time the P-T limit curves are revised based on the results of the vessel material surveillance. SQN's current power-operated relief valve setpoints for LTOP were evaluated with the updated P-T limits to ensure these setpoints provide sufficient margins against overpressure transients. The evaluation show that new setpoints for SQN's Unit 2 LTOP system are not required for the revised 32 EFPY P-T limits. This is

based on the current LTOP setpoints being more restrictive than the setpoints developed for the 32 EFPY P-T limits. In addition, the analyses show that SQN's current vent size of 3.0 square inches is capable of mitigating a LTOP transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration. Accordingly, SQN's current LTOP system setpoints and SQN's current RCS vent size are capable of mitigating LTOP transients.

## 5.0 REGULATORY SAFETY ANALYSIS

In accordance with 10 CFR 50.36, TVA proposes to amend the Sequoyah Nuclear Plant (SQN) Unit 2 Technical Specification (TS) 3/4.4.9.1, "Pressure/Temperature Limits, Reactor Coolant System." The proposed amendment deletes the current figures referenced in Limiting Condition for Operation 3.4.9.1 (Figures 3.4-2 and 3.4-3) and replaces the figures with updated pressure-temperature (P-T) limit curves for reactor coolant system (RCS) heatup and cooldown. The applicability of the updated limits is extended from the current 14.5 Effective Full Power Years (EFPYs) to 32 EFPYs.

### 5.1. No Significant Hazards Consideration Determination

TVA has concluded that operation of SQN Unit 2 in accordance with the proposed change to the TSs, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed revision does not affect plant equipment, test methods or operating practices. The modification to SQN TSs is consistent 10 CFR 50, Appendix G in conjunction with alternative methods provided in American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1." The proposed change continues to provide controls for safe operation within the required limits. The proposed changes do not contribute to events or assumptions associated with postulated design basis accidents (DBA). The proposed revisions

continue to maintain the required safety functions. Accordingly, the probability of an accident or the consequences of an accident previously evaluated is not increased.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The proposed revision is not the result of changes to plant equipment, test methods, or operating practices. The proposed revision to the SQN Unit 2 P-T limits continues to ensure that conservative fracture toughness margins are maintained to protect against reactor pressure vessel failure. In addition, SQN's current setpoints for low-temperature overpressure protection were evaluated and are bounding for the proposed 32 EFPY P-T limits. The updated P-T limits are based on NRC approved methodology in conjunction with alternative methods provided in American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1."

The reactor vessel P-T limits are operational limits and are not considered to be contributors to the generation of postulated accidents. The safety functions of the associated systems remain unchanged and do not affect the assumptions of DBAs. The operational limits continue to be governed within the TSs. Accordingly, the proposed change does not create the possibility of a new or different kind of accident.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

TVA's proposed TS amendment provides revised reactor pressure vessel P-T limits that are within the design capabilities of the pressure control systems for protection of the RCS. The limits are based on conservative design margins that ensure that plant operation is within the design capacity of the reactor vessel materials.

Accordingly, the function of the RCS to provide a fission product barrier is not compromised.

TVA's proposed change to revise P-T limits does not result in a change to system design features. The proposed change does not affect plant conditions that result in precursors to accidents or cause degradation of accident mitigation systems. The plant system safety functions are not altered by the proposed change.

The proposed changes allow plant operation with different P-T limits while continuing to retain conservative margins for assuring integrity of the reactor vessel and the RCS. Consequently, the proposed TS revisions do not significantly reduce the margin of safety.

5.2. Applicable Regulatory Requirements/Criteria

The pressure-temperature (P-T) limits are established by requirements defined in 10 CFR 50, Appendix G, entitled "Fracture Toughness Requirements." These limits are periodically reanalyzed and revised as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. The P-T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. Adjustment of the limits is acceptable if performed in accordance with methodology approved by the NRC. The NRC approved methodology for Westinghouse Electric Company's pressurized water reactor plants is prescribed in Westinghouse Topical Report WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January 1996."

Alternatives to the 10 CFR 50, Appendix G requirements for development of P-T limits are provided by American Society of Mechanical Engineers (ASME) Code Cases (such as Code Case N-640) and must be approved for use. TVA's proposed change to update SQN's Unit 2 P-T limits utilizes ASME Code Case N-640 as documented in WCAP-15321, Revision 1, "Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation, April 2001." This Unit 2 specific analysis updates these

limits within applicable plant design assumptions as stated in the SQN Final Safety Analysis Report (FSAR). These safety analyses demonstrate that SQN's Unit 2 reactor vessel is adequately protected against brittle fracture when operated within these limits.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

## 6.0. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

## 7.0. REFERENCES


1. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

2. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January 1996."

3. WCAP-15321, Revision 1, "Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation, April 2001."

4. TVA letter to NRC dated September 6, 2002, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 Technical Specification (TS) Change No. 00-14, 'Pressure Temperature Limits Report (PTLR) and Request for Exemption From the Requirements of 10 CFR 50, Appendix G.'"

**ENCLOSURE 2**

**TENNESSEE VALLEY AUTHORITY**
**SEQUOYAH NUCLEAR PLANT (SQN)**
**UNIT 2**

**Proposed Technical Specification Changes (mark-up)**

**I. AFFECTED PAGE LIST**

Unit 2

3/4 4-29
3/4 4-30
3/4 4-35

B 3/4 4-6
B 3/4 4-7
B 3/4 4-8
B 3/4 4-11
B 3/4 4-12
B 3/4 4-13
B 3/4 4-14

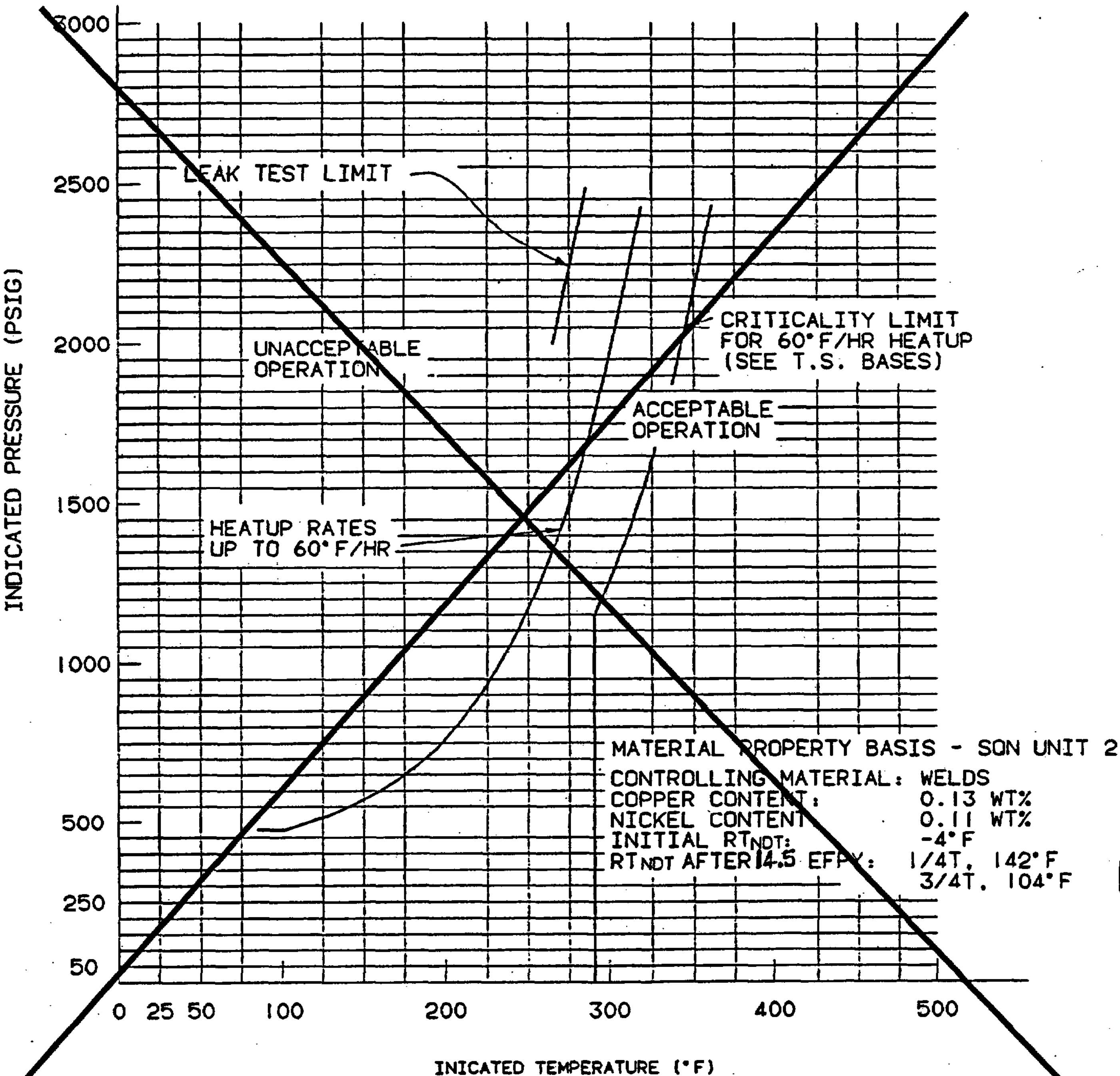
**II. MARKED PAGES**

See attached.

REPLACE FIGURE 3.4-2 WITH ATTACHMENT 1

32

PPLICABLE FOR HEATUP RATES UP TO 60°F/HR
FOR THE SERVICE PERIOD UP TO ~~14.5~~ EFPY. MARGINS OF 60 PSIG AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENT ERRORS.




32

FIGURE 3.4-2 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO ~~14.5~~ EFPY

REPLACE FIGURE 3.4-3 WITH ATTACHMENT 2

32

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO ~~14.5~~ EFPY. MARGINS OF 60 PSIG AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENT ERRORS.

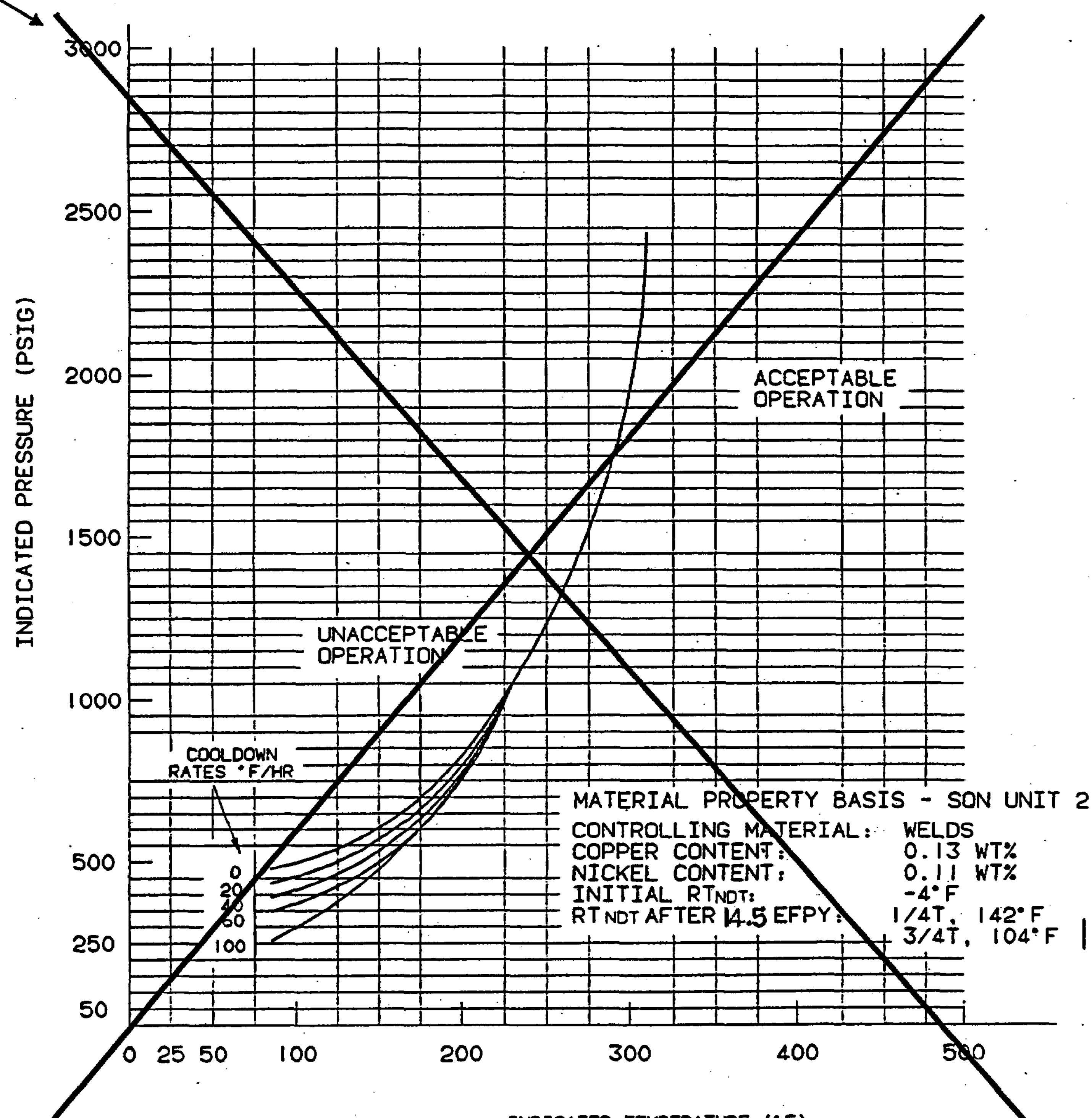



FIGURE 3.4-3 SEQUOYAH UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE UP TO ~~14.5~~ EFPY

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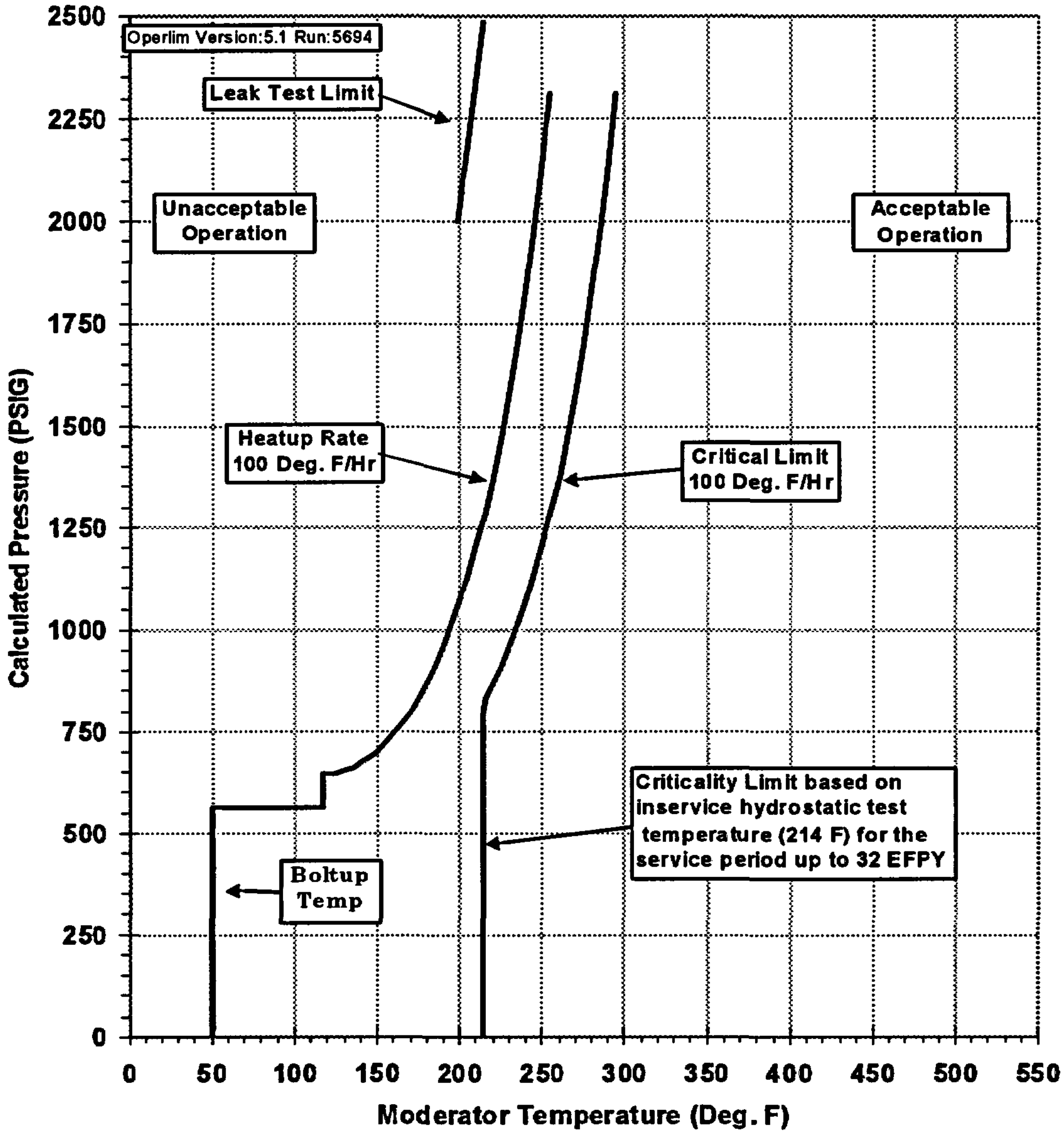
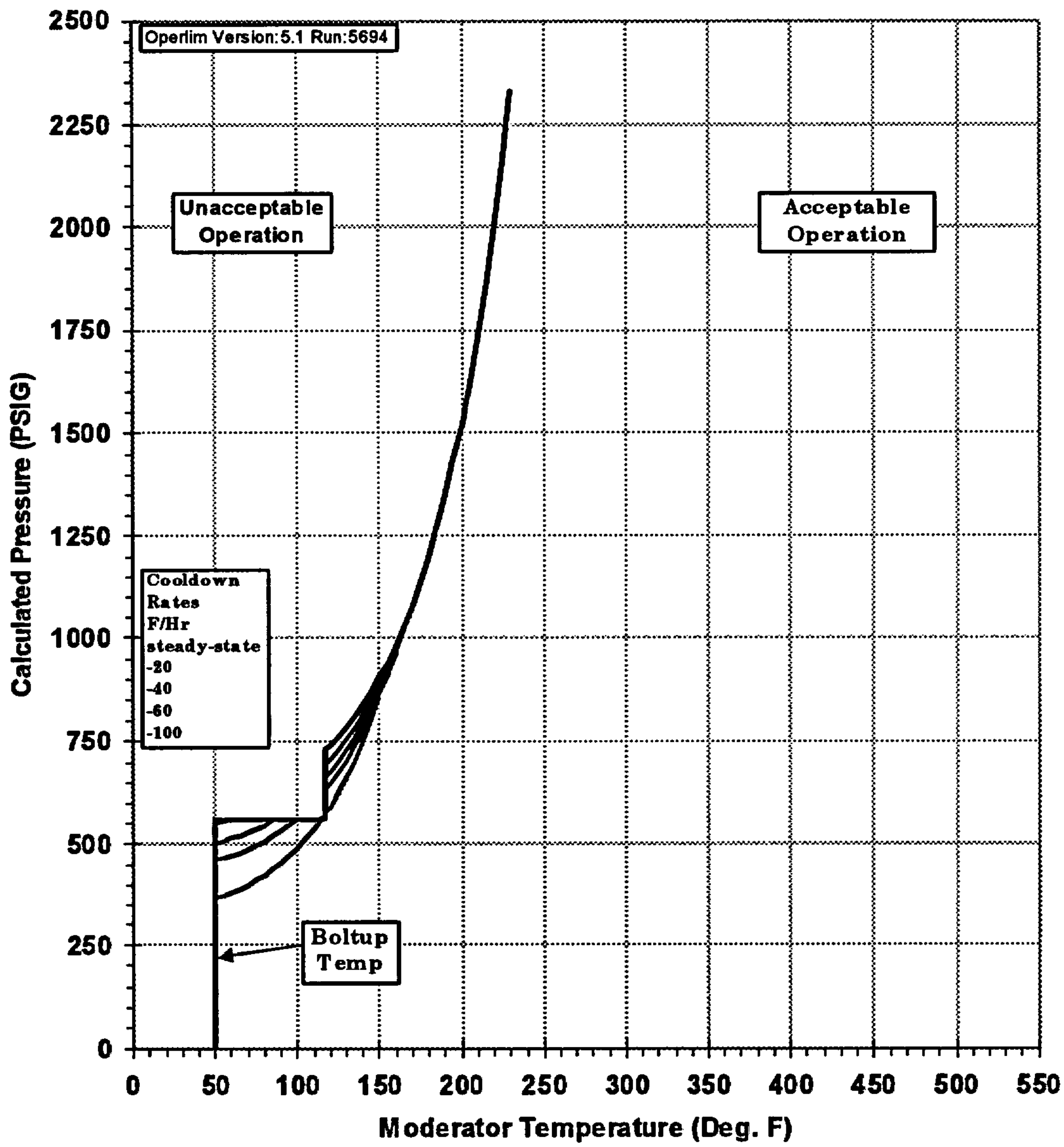
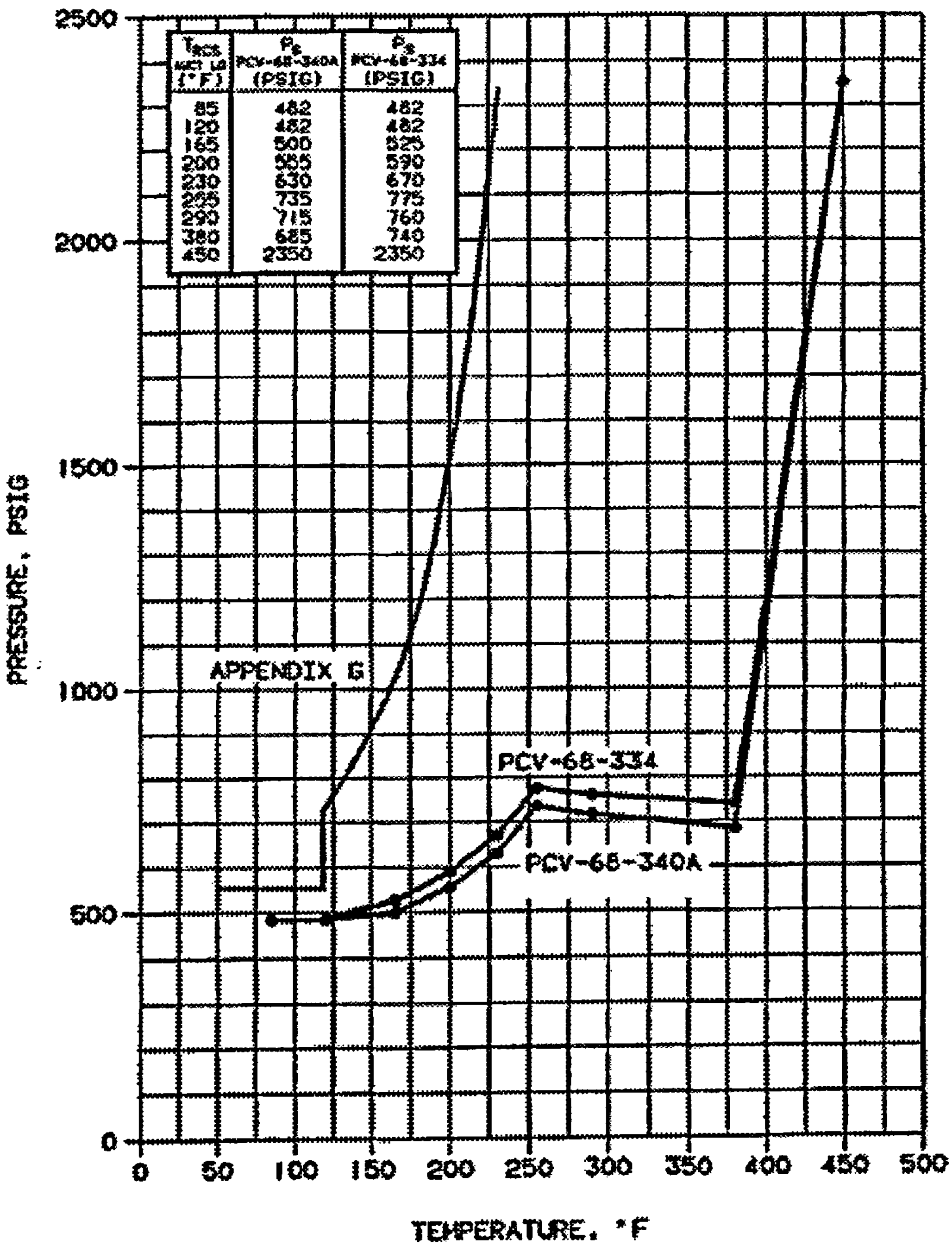
## ATTACHMENT 1



## ATTACHMENT 2






| $T_{RCS}$ WCT LG (°F) | $P_S$ PCV-68-340A (PSIG) | $P_S$ PCV-68-334 (PSIG) |
|---|---|---|
| 85 | 482 | 482 |
| 120 | 482 | 482 |
| 165 | 500 | 525 |
| 200 | 555 | 590 |
| 230 | 630 | 670 |
| 255 | 735 | 775 |
| 290 | 715 | 760 |
| 380 | 685 | 740 |
| 450 | 2350 | 2350 |

$\boxed{32}$

PORV NOMINAL LIFT SETTINGS - APPLICABLE UP TO ~~14.5~~ EFPY

FIGURE 3.4-4

SPECIFIC ACTIVITY (Continued)

Reducing $T_{avg}$ to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

XI

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section ~~III~~, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the first full-power service period.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/ hr and 200°/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 560°F.

PRESSURE/TEMPERATURE LIMITS (Continued)

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

10 CFR 50, Appendix G, addresses metal temperature of the closure head flange and vessel regions. Appendix G states that the minimum metal temperature of the closure flange region should be at least 120 degrees Fahrenheit (F) higher than the limiting $RT_{NDT}$ for this region when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (561 pounds per square inch gauge (psig) for Westinghouse Electric Corporation plants). For SQN, Unit 2, the minimum temperature of the closure flange and vessel flange regions is 117 degrees F since the limiting initial $RT_{NDT}$ for the closure head flange is -13 degrees F (see Table B 3/4.4-1). These numbers (561 psig and 117 degrees F) include a margin for instrumentation error of 10 degrees F and 60 psig. ~~The SQN Unit 2 heat up and cooldown curves shown in Figures 3.4-2 and 3.4-3 are not impacted by this regulation.~~

DELETE

~~The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, and ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to 10 CFR 50 and Appendix G of the 1986 ASME Boiler and Pressure Vessel Code, Section III, Division 1 and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."~~

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Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, $RT_{NDT}$ at the end of ~~16~~ effective full power years of service life. The ~~16~~ EFPY service life period is chosen such that the limiting $RT_{NDT}$ at the 1/4T location in the core region is greater than the $RT_{NDT}$ of the limiting unirradiated material. The selection of such a limiting $RT_{NDT}$ assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

REPLACE WITH INSERT A

~~The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence of the material in question, has been predicted using Regulatory Guide 1.99, Revision 2 and a peak surface fluence of $0.864 \times 10^{19}$ n/cm$^2$ for 16 effective full power years (Reference WCAP-12971, "Heatup and Cooldown Limit Curves for Normal Operation," June 1991. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in $RT_{NDT}$ at the end of 16 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments. The heatup and cooldown limits in WCAP-12971 were based on a core thermal power of 3411 MWt. The curves have been evaluated in WCAP-15725 to be effective for operation through the end of 14.5 EFPY for the uprated core thermal power of 3455 MWt.~~

## INSERT A

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence of the material in question, has been predicted using Regulatory Guide 1.99, Revision 2 and a peak surface fluence of $1.82 \times 10^{19}$ n/cm$^2$ for 32 effective full power years (reference WCAP-15321, Revision 1, "Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," April 2001). The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in $RT_{NDT}$ at the end of 32 EFPYs, as well as adjustments for possible errors in the pressure and temperature sensing instruments. The heatup and cooldown limits in WCAP-15321, Revision 1 were based on a core thermal power of 3411 MWt. The curves have been evaluated in WCAP-15725 to be still effective for operation through the end of 32 EFPYs for the uprated core thermal power of 3455 MWt.

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

DELETE

~~Values of $\Delta RT_{NDT}$ determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. The first capsule will be removed at the end of the first core cycle. Successive capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The heatup and cooldown curves and the low temperature overpressure protection setpoints must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule exceeds the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.~~

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section ~~III~~ XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 ~~and these methods are discussed in detail in WCAP-7924-A.~~ the Summer 1996 Addenda of .

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimenstions of this postulated crack, referred to in Appendix G of ASME ~~III~~ as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, $RT_{NDT}$, is used and this includes the radiation induced shift, $\Delta RT_{NDT}$, corresponding to the end of the period for which heatup and cooldown curves are generated.

PRESSURE/TEMPERATURE LIMITS (Continued)

INSERT B

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, $K_I$, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, $K_{IR}$, for the metal temperature at that time. $K_{IR}$ is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The $K_{IR}$ curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

$K_{IC}$

where $\cancel{K_{IR}}$ is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature $RT_{NDT}$. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq \cancel{K_{IR}} \qquad (2)$$

$K_{IC}$

IC

Where, $K_{IM}$ is the stress intensity factor caused by membrane (pressure) stress.

$K_{It}$ is the stress intensity factor caused by the thermal gradients.

$K_{IC}$ $\cancel{K_{IR}}$ is provided by the code as a function of temperature relative to the $RT_{NDT}$ of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

## INSERT B

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, $K_I$, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, $K_{IC}$, for the metal temperature at that time. $K_{IC}$ is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI" [1 & 2] of the ASME Appendix G to Section XI. The $K_{IC}$ curve is given by the following equation:

$$K_{IC} = 32.2 + 20.734 \exp [0.02(T-RT_{NDT})]$$

PRESSURE/TEMPERATURE LIMITS (Continued)

$\boxed{K_{IC}}$

At any time during the heautp or cooldown transient, ~~$K_{IR}$~~ is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for $RT_{NDT}$, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, $K_{IT}$, for the reference flaw are computed. From Equation (2) the pressure stress intensity factors are obtained and from these the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of ~~$K_{IR}$~~ at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in ~~$K_{IR}$~~ exceeds $K_{IT}$, the calculated allowable pressure during cooldown will be greater than the steady-state value.

$\boxed{K_{IC}}$

PRESSURE/TEMPERATURE LIMITS (Continued)

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

$\boxed{K_{IC}}$

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the ~~$K_{IR}$~~ for the 1/4T crack during heatup is lower than the ~~$K_{IR}$~~ for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and ~~different $K_{IR}$'s~~ $\boxed{K_{IC}\text{'s}}$ for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

PRESSURE/TEMPERATURE LIMITS (Continued)

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

The leak test limit curve shown on Figure 3.4-2 represents the minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curve was determined by methods of Branch Technical Position MTEB 5-2 and 10 CFR 50, Appendix G.

The criticality limit curve shown in Figure 3.4-2 specifies pressure-temperature limits for core operation to provide additional margin during actual power production. The pressure-temperature limits for core operation (except for low power physics tests) require the reactor vessel to be at a temperature equal to or higher than the minimum temperature required for the in-service hydrostatic test, and at least 40 degrees F higher than the minimum pressure-temperature curve for heatup and cooldown. The maximum temperature for the in-service hydrostatic test for the SQN Unit 2 reactor vessel is ~~274~~ degrees F. A vertical line at ~~274~~ degrees F on the pressure-temperature curve, intersecting a curve 40 degrees F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel. 214

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 DELETED